FirstEnergy

Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449-9760

NP-33-98-011-01

Docket No. 50-346

License Number NPF-3

September 9, 1999

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Ladies and Gentlemen:

LER 1998-011-01
Davis-Besse Nuclear Power Station, Unit No. 1
Date of Occurrence - October 14, 1998

Enclosed please find revision one for Licensee Event Report 1998-011. This submittal updates the corrective actions and revises the completion date with regard to implementation of changes to the plant emergency operating procedure. This change of completion date was reviewed with the Senior Resident Inspector. Additional analysis of the occurrence is also provided. The changes are marked with a revision bar in the margin. Please destroy or mark superceded, the previous copies of this LER. This LER is being submitted in accordance with 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(iv).

Very truly yours,

JH Lash

James H. Lash
Plant Manager
Davis-Besse Nuclear Power Station

DLM/dlc

Enclosure

cc: J. E. Dyer, Regional Administrator - USNRC Region III
K. S. Zellers, NRC Senior Resident Inspector DB-1
Utility Radiological Safety Board

IE22

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station in this document. Any other actions discussed in the submittal represent intended or planned actions by Davis-Besse. They are described only as information and are not regulatory commitments. Please notify the Manager - Regulatory Affairs (419-321-8466) at Davis-Besse of any questions regarding this document or associated regulatory commitments.

COMMITMENTS	DUE DATE
1. Several corrective actions have been recommended relative to the human interface design inadequacy for the 4160 volt bus breakers. Final review of the recommended corrective actions will be completed.	November 20, 1998 (Completed)
2. A plant modification to make a permanent change to this time delay circuitry will be initiated.	November 30, 1998 (Completed)
3. Changes to the preventive maintenance procedures and the data packages for CCW flow switches to specify required actions to prevent cycling of valves, will be completed.	January 29, 1999 (Completed)
4. The maintenance procedure for 4160 volt breakers will be revised to include a check of the clearance between the armature buttons and the cover plate.	March 31, 1999 (Completed)
5. A training package was developed and presented by December 7, 1998, to each Operating Crew. This training included information about this event, lessons learned and Plant Operation's management expectations for this type of event.	December 7, 1998 (Completed)
6. An alteration will be completed to DB-OP-02000 and/or the Bases Document for DB-OP-02000 to incorporate lessons learned from this event. Simulator training will be completed utilizing the draft procedure. The revised procedure will be implemented.	October 29, 1999

1205

NRC FORM 366 (6-1998) **U.S. NUCLEAR REGULATORY COMMISSION**

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Davis-Besse Unit Number 1	05000346	1 OF 10

TITLE (4)
Manual Reactor Trip Due to Component Cooling Water System Leak

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	14	1998	1998	-- 011 --	01	09	03	1999		05000
										05000

OPERATING MODE (9): 1

POWER LEVEL (10): 87

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Dale L. Miller, Senior Engineer - Licensing	(419) 321-7264

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	CC	RPD	B295	Y					
X	CC	6	W120	Y					

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (if yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) **(16)**

On October 14, 1998, at 1523 hours, with the Davis-Besse Nuclear Power Station operating in Mode 1, at 87 percent rated thermal power, the plant was manually tripped. Previously on October 14, at 1356 hours, the plant was operating at 100 percent power, when electrical busses D1 and D2 experienced a lockout. As a result of the loss of a condensate pump and several other plant loads powered by these busses, thermal power was reduced to 87 percent. When the lockout occurred, Component Cooling Water (CCW) Pump 1-2 stopped. When CCW Pump 1-1 automatically started, a CCW system leak developed that was determined to be located inside the containment. When CCW pump 1-2 was restarted, the CCW Surge Tank level decreased rapidly. At a surge tank level of 35 inches and decreasing, plant operators tripped the Reactor, and the Reactor Coolant Pumps in accordance with plant procedures. During the trip recovery, a makeup pump failed to start on demand, and a plant overcooling occurred. The cause of the CCW leak was failure of one letdown cooler rupture disk. All of the letdown cooler rupture disks were replaced prior to plant restart. This event was reported to the Nuclear Regulatory Commission (NRC) within four hours via the Emergency Notification System at 1750 hours on October 14, 1998 as an event that resulted in manual actuation of the Reactor Protection System (RPS) in accordance with 10CFR50.72(b)(2)(ii). This report is being submitted in accordance with 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(iv).

Description of Occurrence:

On October 14, 1998, at 1523 hours, with the Davis-Besse Nuclear Power Station (DBNPS) operating in Mode 1, at 87 percent rated thermal power, the plant was manually tripped in accordance with abnormal operating procedure, DB-OP-02523, "Component Cooling Water System Malfunctions." Previously on October 14, at 1356 hours, the plant was operating at 100 percent power, when an electrical bus D1, [Energy Industry Identification System-Function Code: EB-BU] bus D2, [EA-BU] and Station Blackout Diesel Generator (SBODG) [EK-DG] lockout occurred. At the time of this event, Bus Tie Transformer AC, [EA-XFMR] which is also capable of providing a backup power supply to 4160 volt essential bus D1 via circuit breaker AACD1, [EB-BKR] was de-energized for fire protection system deluge [KP-SRNK] testing. Routine preventive maintenance had also been performed on circuit breaker AACD1, and the circuit breaker was ready to be rolled back into its cubicle. At 1355 hours on October 14, 1998, an electrician attempted to install the Westinghouse type DHP circuit breaker into cubicle AACD1. After the breaker had been rolled partially into the cubicle, the electrician noted that the circuit breaker was not completely aligned with the floor rail. The electrician repositioned the breaker, then resumed rolling the breaker into the cubicle. As it was being rolled into the cubicle, the metal breaker frame contacted a terminal screw of a time-over-current relay, [EB-51GS] mounted on the cubicle door, which provides backup ground over-current protection for busses D1 and D2. Accidental contact between the terminal and the breaker, which was grounded via the floor rail, provided a path for current to flow from the DC control power bus to ground via the operating coil of an auxiliary trip relay [EB-511X]. This high speed trip relay, which operates in 8 milli-seconds or less, includes a seal-in circuit that ensures the relay stays in the trip state even if the initiating signal is only momentary. The lockout caused non-essential bus D2 supply breaker ABDD2 [EA-BKR] and D1/D2 bus tie breaker AD110 to open, which de-energized bus D2 and essential bus D1 [EB-BKR]. The SBODG output breaker (AD213) [EA-BKR] was also locked out.

Emergency Diesel Generator (EDG) 1-2 [EK-DG] started on low voltage when the D1/D2 bus lockout occurred at 1356 hours. The EDG output breaker AD101 [EB-BKR] could not close because it was also locked out. At approximately 1401 hours, the EDG was shutdown per procedure because no Component Cooling Water was available to cool the engine.

Due to the lockout, Service Water Pump (SWP) 1-2[BI-P] and Component Cooling Water Pump (CCWP) 1-2 [CC-P] stopped. Secondary system loads were also lost, including Condensate Pump 1-2 [SD-P]. All normal station lighting, which was supplied by bus D2, was lost. A plant power reduction was initiated by Plant Operations personnel due to the loss of Condensate Pump 1-2, with the intent of stabilizing reactor power at a level within the capacity of the two available condensate pumps. The power reduction was stopped at approximately 87 percent power at 1430 hours.

Auxiliary Feedwater Pump (AFP) 1-1 was out of service for the monthly jog test in accordance with DB-SP-03150 prior to the lockout event. Test activities that affect AFP operability were completed and AFP 1-1 was declared operable at 1415 hours on October 14,1998. From 1356 hours until 1415 hours, both auxiliary feedwater trains were inoperable. Auxiliary Feedwater Train 2 was made inoperable as a result of the lockout, which resulted in the plant being in Technical Specification (TS) 3.0.3 for this time period. At the onset of the event, efforts were initiated to restore AFP 1-1 to an operable status, which was accomplished at 1415 hours. Although a plant power reduction was made, the power reduction was not initiated as a result of entry into TS 3.0.3.

Description of Occurrence: (Continued)

Loss of power to busses D1 and F1 placed the plant into the Actions for the TS listed below:

TS 3.1.2.4	Reactivity Control Systems - Makeup Pumps
TS 3.3.2.2	Steam and Feedwater Rupture Control System Instrumentation
TS 3.7.1.2	Auxiliary Feedwater System
TS 3.7.1.7	Motor Driven Feedwater Pump System
TS 3.8.1.1	Electrical Power Systems - A.C. Sources
TS 3.8.2.1	Onsite Power Distribution Systems - A.C. Distribution
TS 3.8.2.3	Electrical Power Systems - D.C. Distribution

As a result of the loss of the 4160 volt D1 electrical bus at 1356 hours on October 14, 1998, several TSs, listed below, were entered because both the normal and emergency power sources for the affected equipment were inoperable. As a result, the unit was required to satisfy the Limiting Condition for Operation of TS 3.0.5.

TS 3.3.3.1	Radiation Monitoring Instrumentation
TS 3.4.6.1	Reactor Coolant System Leakage Detection System
TS 3.5.2	Emergency Core Cooling System Subsystems
TS 3.6.2.1	Containment Spray Systems
TS 3.6.2.2	Containment Cooling Systems
TS 3.6.3.1	Containment Isolation Valves
TS 3.6.4.1	Combustible Gas Control Hydrogen Analyzers
TS 3.6.4.3	Containment Hydrogen Dilution System
TS 3.6.4.4	Containment Hydrogen Purge System
TS 3.6.5.1	Shield Building Emergency Ventilation System
TS 3.7.3.1	Component Cooling Water System
TS 3.7.4.1	Service Water System
TS 3.7.6.1	Control Room Emergency Ventilation System
TS 3.9.12	Storage Pool Ventilation

At 1512 hours on October 14, 1998, electrical busses D1 and F1 were re-energized, which returned the normal power supply affecting the TS systems or components listed above, and these TSs were exited. The action requirements of TS 3.0.5 were satisfied.

Prior to the bus D1/D2 lockout, CCWP 1-2 was operating and supplying non-essential CCW loads inside Containment. Troubleshooting was in progress on the CCWP 1-1 discharge flow indicating switch FIS1422D [CC-FIS], which was isolated and drained. Both Reactor Coolant System (RCS) letdown coolers [AB-HX] were in-service. When the bus D1/D2 lockout occurred, CCWP 1-2 tripped. When low pump discharge flow was sensed for CCWP 1-2, a start signal was provided for CCWP 1-1 and open signals were provided to the Loop 1 non-essential isolation valves [CC-ISV]. After CCWP 1-2 tripped and CCWP 1-1 started, the Loop 1 non-essential valves began to open after a thirty second time delay. When CCWP 1-2 tripped, close signals were sent to the Loop 2 non-essential isolation valves. However, the Loop 2 non-essential valves could not close due to the D1/D2 bus lockout.

During the 30 second time delay for the Loop 1 non-essential isolation valves to begin stroking open, no CCW flow was provided to the RCS letdown coolers. Hot reactor coolant flowing through the letdown coolers heated the CCW in the coolers to saturation conditions which caused steam to form in the coolers. When CCWP 1-1 started and re-initiated flow to the letdown coolers, introduction of sub-cooled CCW caused the steam pockets to collapse. The resultant short duration pressure spike

Description of Occurrence: (Continued)

damaged one of the two rupture disks [CC-RPD] on Letdown Cooler 1-1 (PSE 3761). Alarms received for operation of the Containment Normal Sump Pump [NH-P] and low level in the CCW Surge Tank [CC-TK] were correlated, which indicated that a leak of an estimated 2 to 5 gpm had started in Containment from the CCW system.

The breaker being installed into D1 bus cubicle AACD1 was removed from the cubicle. After assessment of the lockout and the status of the electrical distribution system by Plant Maintenance, Operations and Engineering personnel, the lockout was reset, and restoration of electrical busses was commenced. When 480 volt essential bus F1 was re-energized at 1512 hours, power was restored to the CCW Loop 2 non-essential isolation valves. The loop 2 non-essential isolation valves started to cycle open and closed because of an open signal from FIS1422D and a close signal from the breaker interlocks. The valves continued to cycle until CCWP 1-2 was started.

At 1517 hours, SWP 1-2 was restarted followed by the restart of CCWP 1-2 at 1523 hours. When the CCWP 1-2 was started, the CCW Surge Tank level decreased rapidly. At the level of 35 inches and decreasing, the Control Room Senior Reactor Operator instructed the Reactor Operators to trip the Reactor, and to trip the Reactor Coolant Pumps (RCPs) [AB-P] in accordance with DB-OP-02523, "Component Cooling Water System Malfunctions." Plant operators entered the plant emergency procedure DB-OP-02000. The Control Rod Drive Trip Breakers [AA-BKR] opened and all control rods inserted on the reactor trip, as designed. Tripping the RCPs resulted in an actuation of Auxiliary Feedwater (AFW) [BA] from the Steam Feedwater Rupture Control System (SFRCS) [JB] and the establishment of natural circulation core cooling. Natural circulation conditions were fully developed approximately four minutes after the RCPs were tripped.

Following the Reactor trip, operators attempted to start Makeup Pump 1-2 [CB-P] in accordance with procedure. However, the pump did not start.

The steam generator (SG) outlet pressures increased due to the closing of the main turbine stop valves. The turbine bypass valves (TBVs) and the atmospheric vent valves (AVVs) opened and the main steam safety valves (MSSVs) lifted in response to the increasing secondary system pressure. The MSSVs, and the AVVs closed as SG outlet pressure decreased. The TBVs throttled closed as they attempted to control SG outlet pressure at the post-trip setpoint of approximately 995 psig. Following the reactor trip, MSSV SP17B7 [SB-RV] was identified to be not fully closed. Main steam pressure was manually reduced to 920 PSIG, per procedure DB-OP-02000, to reseat MSSV SP17B7. The MSSV reseated and SG pressures stabilized. The plant operators determined that this minor overcooling event had been terminated based on the initial SG pressure response. The magnitude and the rate of this overcooling was within the TS limits for cooldown of the RCS.

Although SG pressure initially recovered within a few minutes, SG pressures and RCS temperature started to decline. The plant operators were aware of the potential for overcooling due to steam production being less than the steam loads for the equipment in service. The RCS cooldown rate was initially determined to be less than 50 degrees Fahrenheit per hour. Since the safety Parameter Display System was not available, the RCS cooldown rate was determined by manual calculation.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Description of Occurrence: (Continued)

Actions proceeded to investigate and recover from the loss of CCW to containment, and recover electrical loads that were lost. Plant operators initiated actions to reduce secondary system steam loads to terminate the overcooling. Prior to the event, the Auxiliary Boiler was out of service for maintenance and code safety valve testing. Plant operators discussed the status of the Auxiliary Boiler with the Plant Engineer who was supervising the code safety valve testing on the Auxiliary Boiler.

When it was apparent to the Shift Supervisor that the efforts to reduce secondary system steam loads would not occur soon enough, the Auxiliary Boiler would not be available, and that the cooldown rate had increased to a rate of 65 degrees Fahrenheit per hour, he directed re-entry into the Overcooling Section of DB-OP-02000 and the manual initiation of SFRCS. As plant operators were being directed to manually initiate and isolate the SFRCS, an automatic Low SG Generator Pressure Trip on SG 2, SFRCS Actuation Channel 2, was received at 620 psig. This automatic action occurred approximately 50 minutes after the MSSV was reseated. Plant operators then manually actuated and isolated the SFRCS. Steam generator pressures increased. Based on the Shift Supervisors direction, auxiliary feedwater was re-aligned to receive steam from, and to supply auxiliary feedwater, to its respective Steam Generator.

Plant conditions were then stabilized in Mode 3. Plant operators made preparations to restore CCW to the Containment while leaving CCW to the letdown coolers isolated. At 1712 hours, CCW was restored to the Containment header to provide cooling for the Control Rod Drives, and RCPs. Shortly thereafter, RCPs 2-2 and 1-2 were started, restoring forced RCS cooling flow.

Both AFPs inoperable is reportable to the Nuclear Regulatory Commission (NRC) in accordance with 10CFR50.73(a)(2)(i)(B) as a condition prohibited by the plant TSs. When CCW was lost for the Reactor Coolant Pumps, the Reactor was manually tripped per plant procedures. This is reportable to the NRC in accordance with 10CFR50.72(b)(2)(ii) within four hours, and the NRC was notified of this event via the Emergency Notification System (ENS) at 1750 hours on October 14, 1998, as an event that resulted in manual actuation of the Reactor Protection System (RPS). This report is being submitted in accordance with 10CFR50.73(a)(2)(iv) as an event that resulted in manual actuation of the RPS and 10CFR50.73(a)(2)(i)(B).

An evaluation of the CCW surge tank level decrease, coupled with cycling of the non-essential CCW isolation valves produced a conclusion that a common mode failure mechanism existed. This was reported to the NRC via the ENS at 2048 hours on October 14, 1998, in accordance with 10CFR50.72(b)(2)(iii)(A) as a condition that alone could have prevented the fulfillment of a safety function of a system needed to shutdown the reactor and maintain it in a safe shutdown condition. Further analysis of event data demonstrated that the CCW containment isolation valves (CC1411A and CC1411B) functioned as designed to isolate Letdown Cooler 1-1 within 10 seconds, on low CCW surge tank level, to terminate the leak, maintain CCW system inventory and maintain the required net positive suction head to the CCW pumps. Non-essential valve cycling was determined to not affect the safety function of the CCW system. As a result, on October 17, 1998, the NRC was notified via the ENS that the notification made on October 14, 1998, at 2048 hours was being withdrawn.

Apparent Cause of Occurrence:

The precursor event to the CCW Letdown Cooler rupture disk failure was a lockout of the D1 and D2 electrical busses. As a 4160 volt breaker was being installed into D1 bus cubicle AACD1, plant personnel noticed that the breaker was not properly aligned with the cubicle floor rails. After plant personnel repositioned the breaker and then rolled the breaker into the cubicle, the metal breaker frame contacted a terminal screw of a ground overcurrent relay, which caused current to flow from the DC control power bus to ground. The apparent cause for this occurrence is an inadequate design layout of the relays mounted on the cubicle door which created a human interface inadequacy. The configuration of the relays mounted on the switchgear door provides insufficient clearance between the circuit breakers and the exposed relay terminals. Some bus cubicles, including cubicle AACD1, have two vertical columns of protective relays mounted on their door. For this configuration, when the door is held in its maximum open position, the clearance between the breaker frame and the closest relay terminal is estimated to be only one inch. Contributing to the cause of occurrence was a failure of plant personnel to evaluate the conditions noted during the activity and allowing the breaker frame to contact the relay terminal.

The D1/D2 bus lockout caused a loss of the operating CCW pump which was followed by failure of letdown cooler rupture disk PSE3761. The primary cause of this event was a design configuration that allowed the formation and subsequent collapse of steam voids within a heat exchanger designed with rupture disks for overpressure protection. Letdown flow through the coolers was approximately 25 gpm per cooler at a temperature of approximately 558 degrees Fahrenheit. Normal CCW flow is approximately 390 gpm per cooler at 80 psig and 85 degrees Fahrenheit. Following a loss of the operating CCW Pump, CCW pressure drops to approximately 20 psig and total CCW pump flow drops to 1000 gpm before the standby pump starts. Thirty seconds after the standby pump starts, the associated non-essential isolation valves will begin to open to re-establish flow to the Letdown Coolers. This design would result in a total loss of CCW flow to the Letdown Coolers for approximately 30 seconds. During this time, letdown flow at 25 gpm and 558 degrees continued to both coolers, causing the CCW within the coolers to boil and form steam voids within the coolers. Saturation temperature for water at 20 psig is approximately 258 degrees Fahrenheit. The bulk CCW temperatures noted downstream of the Letdown Coolers following flow recovery were 240 degrees Fahrenheit. The computer points monitoring these temperatures have a thirty second scan time and actual peak temperatures within the cooler would exceed bulk temperatures downstream due to mixing action, supporting the conclusion that saturation temperatures were reached within the coolers. When CCW flow was recovered, colder water was forced into the coolers causing the steam voids to collapse, resulting in a water hammer type event. The pressure pulse caused by the steam void collapse was sufficient to damage the inlet rupture disc on Letdown Cooler 1-1 resulting in the initial 2-5 gpm leak. In 1993, a plant modification installed the current rupture disks with a 250 PSI setpoint. The analysis and modification didn't recognize the potential for higher pressure pulses due to steam voiding during the transfer from an operating CCW pump to the standby pump.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Davis-Besse Unit Number 1	05000346	1998	-- 011 --	01	7 OF 10

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Apparent Cause of Occurrence: (Continued)

Contributing to the cause of this event was that the troubleshooting activities for FIS 1422D locked in an open signal to the Loop 2 non-essential isolation valves. When power was restored to the Loop 2 non-essential isolation valves, these valves started to cycle open and closed, and continued to cycle until CCWP 1-2 was started. This was a distraction and did not initially contribute to the event. However, when CCW Pump #2 was restarted, this cycling resulted in a hydraulic pressure pulse from the pump start being transmitted to the Letdown Cooler rupture disc previously damaged, which caused the size of the CCW leak to increase significantly. The effects of FIS 1442D troubleshooting activities on the system were discussed with Plant Operations prior to beginning work. This work was commenced because open signals were being sent to valves already aligned open and that the troubleshooting activities would not effect the interlocks that would transfer CCW loads to the standby pump following a loss of the operating CCW Pump. The potential to cause the non-essential isolation valves on Loop 2 to cycle following a loss of the operating pump was not identified.

Failure of MUP 1-2 to start is attributed to an inadequate gap between the armature buttons of the breakers' anti-pumping relay and the relay cover plate. Bench testing of the anti-pump relay removed from the breaker determined that the slightest pressure on the armature buttons caused a high resistance or open contact in series with the breaker spring release circuit coil. This high resistance or open contact would prevent the breaker from closing because the spring release circuit could not pick up. With the relay installed in the breaker, the relay cover plate was observed to be in contact with the relay's armature buttons. It was concluded that there was sufficient contact between the cover plate and armature buttons to prevent the spring release circuit from picking up.

Overcooling of the RCS following the reactor trip was a result of the steam demands on the Main Steam system to supply the secondary systems being greater than the amount of decay heat available for steam production. The Auxiliary Boiler was not available to supply the steam loads in the secondary systems, due to code safety valve testing. The RCPs had been shutdown due to a loss of CCW to containment which removed approximately 16 megawatts thermal heat input. The shutdown of the RCPs then resulted in automatic initiation of the AFPs, adding additional steam loads. During the initial entry into the Overcooling Section of DB-OP-02000, "RPS, SFAS, SFRCS Trip or SG Tube Rupture," SG pressure was reduced to reseat MSSV SP17B7. The reduction in pressure did result in the MSSV reseating. An incorrect assumption was made by the plant operators that the overcooling event had been terminated based on the initial SG pressure response. As a result, the Overcooling Section of the procedure was exited and Supplemental Actions were continued. Although pressure did initially recover, within a few minutes, SG pressures and RCS temperature started to decline. If the Overcooling Section of the procedure had been continued, operators would have been directed to initiate and isolate the SFRCS, which would have terminated the overcooling event. When the plant operators recognized that the overcooling was still in progress, they delayed re-entering the Overcooling Section of the procedure to focus on other plant conditions. The philosophy use document for DB-OP-02000 states that emergency procedure actions take priority over most abnormal procedure actions. Early actuation and isolation of the SFRCS would have terminated the overcooling allowing resources to be focused on dealing with the other plant problems.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Analysis of Occurrence:

This event had no safety significance to the health and safety of the public. Upon initiation of the manual Reactor trip, the Reactor Protection System and the Control Rod Drive Trip Breakers functioned properly and all Control Rods inserted as designed. After the RCPs were tripped, the Steam and Feedwater Rupture Control System was actuated, both AFPs started, and natural circulation flow was developed in the RCS as expected. The Turbine Bypass Valves, Atmospheric Vent Valves and Main Steam Safety Valves functioned to cool the RCS as required.

Although the combination of events that occurred relative to the CCW system rupture disk failure caused the Reactor to be manually tripped, the CCW leak in the non-essential header was automatically isolated on low CCW Surge Tank level as designed. The safety function of the CCW system was preserved and the CCW system was available to provide its safety-related cooling functions as designed. A review of heat exchangers serviced by the CCW system determined that there are no other heat exchangers designed with rupture disks for over-pressure protection. A review of other cooling water systems (Service Water, Turbine Plant Cooling Water, and Chilled Water) also identified no heat exchangers with rupture disks installed. There were also no other heat exchangers identified with a large heat gradient between the tube and shell sides as the letdown coolers, making these coolers the most susceptible to steam voiding on a loss of CCW flow.

Failure of MUP 1-2 to start was noteworthy because the required gap between the breaker anti-pump relay armature buttons and the protective cover was not previously identified as a possible cause of the breaker's failure to close. This was the first documented failure of a breaker to close because of this deficiency. Previously documented closure failures were traced to either the latch check switch or the motor cutoff switch. The complete population of 4160 volt breakers, which is 51 operational breakers and 3 spare breakers, was inspected. This inspection revealed three other breakers with no clearance between the armature buttons and the cover plate, which could have potentially affected their ability to close on demand. Two of these breakers, AD111 for High Pressure Injection Pump 1-2 and AD301 for the SBODG, were in service. The third breaker, for CCWP 1-3, was not in service.

The overcooling event had minimal safety significance. The maximum cooldown rate calculated from the plant post-trip data was approximately 90 degrees Fahrenheit per hour. The maximum allowable RCS cooldown rate limit provided in TS 3.4.9.1, Reactor Coolant System Pressure - Temperature Limits, is 100 degrees Fahrenheit per hour for RCS temperatures greater than or equal to 270 degrees Fahrenheit. The RCS temperature was maintained greater than 270 degrees Fahrenheit during this event. Evaluation of the procedures and systems used for mitigation of the overcooling indicated they are capable of performing their intended function.

The short time, 19 minutes, that both AFPs were inoperable at the time of the lockout had no safety significance. The AFW system was not required to perform any safety function during the time that the AFPs were inoperable. When the lockout occurred, the monthly jog test of AFP 1-1 was nearly complete. The portion of the test that affected AFP 1-1 operability had been satisfactorily completed and the AFP was functional, which allowed AFP 1-1 to be expeditiously declared operable. The AFW Train 2 was made inoperable during the lockout due to loss of power to some train 2 motor operated valves. The AFP 1-2 remained functional and would have been able to feed its respective steam generator. When the Reactor and RCPs were tripped as a result of the CCW leak, both AFPs started and the AFW system successfully performed its safety function.

Analysis of Occurrence: (Continued)

Main Steam Safety Valve, SP17B7, opened as required during the event. However, the valve did not reseat until main steam pressure was reduced to 920 PSIG. A setpoint check using a hydroset conducted on October 18, 1998, showed the valve lifted within one percent of its 1050 PSIG setpoint, two times. No setpoint adjustments were required. A minor over-cooling of the RCS resulted from reducing the Steam Generator outlet steam pressure approximately 75 PSIG below the normal post-trip setpoint of 995 PSIG. The magnitude and rate of this overcooling was within the TS limit for cooldown of the RCS.

Corrective Actions:

After the D1/D2 bus lockout occurred, the breaker being installed into D1 bus cubicle AACD1 was removed from the cubicle. After assessment of the lockout and the status of the electrical distribution system by Plant Maintenance, Operations and Engineering personnel, the lockout was reset, and restoration of electrical busses was completed. After review of this occurrence within the DBNPS corrective action program, several corrective actions have been recommended relative to the inadequate human interface design inadequacy for the 4160 volt bus breakers. Final review of the recommended corrective actions will be completed by November 20, 1998.

Following the failure of the letdown cooler rupture disks, all four letdown cooler rupture disks were replaced prior to plant startup. A plant modification (98-0050) was initiated to replace the existing letdown cooler rupture disks with another means of overpressure protection. A Temporary Modification (98-0037)was implemented to reduce the time delay for opening the non-essential isolation valves from 30 seconds to 5 seconds to reduce the potential for steam void formation in the letdown coolers. A plant modification to make a permanent change to this time delay circuitry will be initiated by November 30,1998. Changes to the preventive maintenance procedures and the data packages for CCW flow switches to specify required actions to prevent cycling of valves, will be completed by January 29, 1999.

When the cause of MUP 1-2 failure to start was determined to be inadequate clearance between the armature buttons and the cover plate, maintenance work activities were completed to inspect all 4160 volt breakers. Repairs were made to the three breakers containing anti-pump relays that did not demonstrate adequate clearance between the armature buttons and the cover plate. The maintenance procedure for 4160 volt breakers will be revised to include a check of the clearance between the armature buttons and the cover plate. This procedure revision will be completed by March 31, 1999.

Actions relative to the overcooling event are as follows:

- A training package was developed and presented by December 7, 1998, to each Operating Crew. This training included information about this event, lessons learned and Plant Operation's management expectations for this type of event.
- A temporary change was made to DB-OP-02000 that provided interim guidance to plant operators for overcooling events resulting from secondary steam demands exceeding primary heat production. This change was included in required reading.
- An alteration will be completed to DB-OP-02000 and/or the Bases Document for DB-OP-02000 to incorporate lessons learned from this event. Simulator training will be completed utilizing the draft procedure. The revised procedure will be implemented by October 29, 1999.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Davis-Besse Unit Number 1	05000346	1998	-- 011 --	01	10 OF 10

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Failure Data:

Within the last 3 years there have been 4 reactor trips reported in accordance with 10CFR50.72(b)(2)(ii) and 10CFR50.73(a)(2)(iv). Two LERs involved a manual reactor trip. A resin breakthrough for Makeup and Purification Demineralizer 1-3 caused a loss of letdown capability which precipitated a manual reactor trip on high pressurizer level reported in LER 98-002. Inadvertent closure of a main feedwater regulating valve during testing caused a loss of main feedwater which lead to the manual reactor trip reported in LER 98-010. Two LERs involved automatic reactor trips. An automatic reactor trip occurred when an inadvertent actuation of the main transformer fire protection deluge system caused a main generator lockout as reported in LER 97-010. A tornado caused a loss of off-site power and a turbine-generator load rejection which lead to the reactor trip reported in LER 98-006. There have been no manual reactor trips in the last 3 years as a result of a loss of CCW to the RCPs.

NP-33-98-011-0

PCAQR 98-1854
PCAQR 98-1855
PCAQR 98-1857
PCAQR 98-1858
PCAQR 98-1859
PCAQR 98-1860
PCAQR 98-1868
PCAQR 98-1884

CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9909210094 DOC.DATE: 99/09/09 NOTARIZED: NO DOCKET #
FACIL:50-346 Davis-Besse Nuclear Power Station, Unit 1, Toledo Edi 05000346
AUTH.NAME AUTHOR AFFILIATION
LASH,J.H. Centerior Energy
RECIP.NAME RECIPIENT AFFILIATION
Records Management Branch (Document Control Desk)

SUBJECT: Forwards rev one for LER 1998-001,updating corrective actions & revising completion date re implementation of changes to plant emergency operating procedure.List of commitments attached.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 2+10
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	LPD3-2	1	1	BAILEY,S	1	1
INTERNAL:	ACRS	1	1	FILE CENTER	1	1
	NRR/DIPM/IOLB	1	1	NRR/DRIP/REXB	1	1
	NRR/DSSA/SPLB	1	1	RES/DET/ERAB	1	1
	RES/DRAA/OERAB	1	1	RGN3 FILE 01	1	1
EXTERNAL:	L ST LOBBY WARD	1	1	LMITCO MARSHALL	1	1
	NOAC POORE,W.	1	1	NOAC QUEENER,DS	1	1
	NRC PDR	1	1	NUDOCS FULL TXT	1	1

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LIS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTR DESK (DCD) ON EXTENSION 415-2083

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 16 ENCL 16